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Docket No. 50-366

HL-6184

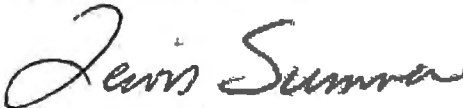
U.S. Nuclear Regulatory Commission
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Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Sudden Closure of Main Steam Line Isolation Valve Causes
Pressure Increase and Reactor Scram on APRM High Flux

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a sudden closure of a main steamline isolation valve which caused a pressure increase and reactor scram on APRM high flux.

Respectfully submitted,


H. L. Sumner, Jr.

CLT/eb

Enclosure: LER 50-366/2001-003

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U.S. Nuclear Regulatory Commission, Washington, D.C.
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IE22

LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME Edwin I. Hatch Nuclear Plant - Unit 2	2. DOCKET NUMBER 05000-366	3. PAGE 1 OF 4
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4. TITLE Sudden Closure of Main Steam Line Isolation Valve Causes Pressure Increase and Reactor Scram on APRM High Flux
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5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
12	25	2001	2001	003	0	02	14	2002		05000
									FACILITY NAME	DOCKET NUMBER(S)
										05000

9. OPERATING MODE 1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)									
10. POWER LEVEL 100	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)			
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)			
	20.2203(a)(1)		50.36(c)(1)(i)(A)		X 50.73(a)(2)(iv)(A)		73.71(a)(4)			
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)			
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER			
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A			
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)					
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)					
20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)						
20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)						

12. LICENSEE CONTACT FOR THIS LER									
NAME Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch						TELEPHONE NUMBER (include Area Code) (912) 367-7851			

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT										
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SB	SHV	R344	Yes						

14. SUPPLEMENTAL REPORT EXPECTED						15. EXPECTED SUBMISSION DATE		MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X				NO					

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On 12/25/2001 at 1819 EST, Unit 2 was in the Run mode. At that time, the reactor scrambled on Average Power Range Monitor high neutron flux caused by a rapid increase in reactor pressure vessel pressure. Pressure increased quickly as a result of the unexpected and sudden closure of main steam line isolation valve 2B21-F028B. The closure of the main steam line isolation valve isolated one of the four main steam lines. Although the flow rates in the remaining three steam lines increased to compensate partially for the isolated line, the sudden isolation of one line was sufficient to cause reactor vessel pressure to increase from a nominal value of 1035 psig to 1041.2 psig within 0.3 seconds. This rapid rate of change in pressure caused reactor power to increase to 120.5 percent rated thermal power and the reactor to scram on high neutron flux level. Following the scram, water level decreased due to void collapse from the rapid reduction in power resulting in closure of Group 2 primary containment isolation valves. Level reached a minimum of 33.5 inches below instrument zero, a level not low enough to initiate other protective actions. Therefore, no systems other than the Group 2 primary containment isolation valves actuated or were required to actuate. The Reactor Feedwater Pumps restored level to its pre-event value of approximately 36 inches above instrument zero within 30 seconds of the scram. Reactor pressure reached its maximum value of 1048.2 psig less than one second after the scram. It decreased thereafter and was maintained below 975 psig by the main turbine bypass valves. No safety/relief valves lifted nor were any required to lift to reduce pressure.

This event was the result of component failure caused by high-cycle fatigue. The stem in valve 2B21-F028B failed completely, causing the valve to close and reactor vessel pressure to increase. Corrective actions include replacing the stem and determining the feasibility and cost of options to reduce or eliminate stem vibration.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System codes appear in the text as (EIIIS Code XX).

DESCRIPTION OF EVENT

On 12/25/2001 at 1819 EST, Unit 2 was in the Run mode. At that time, the reactor scrammed on Average Power Range Monitor (APRM, EIIIS Code IG) high neutron flux after reactor power had increased to approximately 120.5 percent rated thermal power as a result of a rapid increase in reactor pressure vessel pressure. Pressure increased quickly as a result of the unexpected and sudden closure of main steam line isolation valve (EIIIS Code SB) 2B21-F028B. The closure of the main steam line isolation valve isolated one of the four main steam lines (EIIIS Code SB). Although the flow rates in the remaining three steam lines increased to compensate partially for the isolated line, the sudden isolation of one steam line was sufficient to cause reactor vessel pressure to increase from a nominal value of 1035 psig to 1041.2 psig within 0.3 seconds. This rapid rate of change in pressure caused reactor power to increase to 120.5 percent rated thermal power within the same 0.3-second period and the reactor to scram on high neutron flux level per design.

Following the automatic reactor scram, vessel water level decreased due to void collapse from the rapid reduction in power. Water level reached a minimum of 33.5 inches below instrument zero (approximately 125 inches above the top of the active fuel) resulting in closure of the Group 2 primary containment isolation valves (EIIIS Code JM). Water level, however, did not decrease to the actuation setpoint for any other protective action system; therefore, no systems other than the Group 2 primary containment isolation valves actuated or were required to actuate.

The Reactor Feedwater Pumps (EIIIS Code SJ) rapidly recovered reactor vessel water level, restoring level to its pre-event value of approximately 36 inches above instrument zero within 30 seconds of the scram.

Reactor pressure reached its maximum value of 1048.2 psig 0.6 seconds after the scram. It decreased thereafter and was maintained below 975 psig by the main turbine bypass valves. No safety/relief valves lifted nor were any required to lift to reduce pressure.

CAUSE OF EVENT

This event was the result of component failure. Specifically, the stem in main steam line isolation valve 2B21-F028B failed completely from high-cycle fatigue, causing the stem disc (pilot valve) to fall to the closed position. Failure initiation was in the root region of the first thread at the disc-end of the stem. When the stem disc closed, differential pressure forces on the main valve disc (poppet) caused it to close suddenly. The sudden closing of the main steam isolation valve caused reactor vessel pressure to increase from a nominal value of 1035 psig to 1041.2 psig within 0.3 seconds. This rapid rate of change in pressure caused reactor power to increase to 120.5 percent rated thermal power within the same 0.3-second period and the reactor to scram on high neutron flux level per design.

The reason the main steam line isolation valve stem failed due to high-cycle fatigue could not be determined conclusively. The available data support no definitive conclusions regarding the causes of the stem failure. High-cycle fatigue occurs when the number of cycles and level of stress exceed the endurance limit of the failed

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material. Poor surface conditions and degradation of material condition can reduce the stem material's endurance limit to the point that normal cyclic loading would be sufficient to result in fatigue failure. Conversely, cyclic loading stresses and frequency could change such that the expected material endurance limit would be exceeded. The number of cycles and/or the level of stress experienced by isolation valve 2B21-F028B may be different from other isolation valves whose stems have not failed. Also, the stem material's endurance limit may be different: either it changed while the stem was in service (material condition) or it was reduced by a defect (stress riser) in this stem or both. There is insufficient evidence, however, to determine to what extent, if any, these factors contributed to the high-cycle fatigue failure.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv)(A) because of the unplanned actuation of reportable systems. Specifically, the reactor protection system (EIS Code JC) actuated on APRM high neutron flux. Group 2 primary containment isolation valves closed as a result of the expected reactor vessel water level decrease following the scram.

Two isolation valves are welded in a horizontal run in each of the four main steam lines. Each of the main steam line isolation valves is a 24-inch, Y-pattern, globe valve. The main valve disc is attached to the lower end of the stem and moves in guides at a 45-degree angle from the inlet pipe. Normal steam flow and higher inlet pressure tend to close the main valve disc. A stem disc attached to the end of the valve stem closes a small pressure-balancing hole in the main disc. When the pressure-balancing hole is open, it acts as a pilot valve to relieve these differential pressure forces on the main disc thereby allowing it to open.

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (EIS Code IG) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these local power range monitor signals to provide a continuous indication of average reactor power from a few percent to greater than rated thermal power. The APRM high neutron flux function is capable of generating a reactor protection system trip signal in sufficient time to prevent fuel damage or excessive reactor coolant system pressure.

In this event, the reactor scrammed on Average Power Range Monitor high neutron flux resulting from a rapid increase in reactor pressure vessel pressure. Pressure increased quickly as a result of the unexpected and sudden closure of main steam line isolation valve 2B21-F028B. All systems functioned as expected and per their design given the core thermal power, water level, and pressure transients caused by this event. Fuel cladding integrity was not jeopardized because of the rapid response of the APRMs to the neutron flux increase. This response resulted in a reactor scram before the increased energy from the fuel pellets could be transferred fully to the metal cladding. Additionally, reactor vessel water level was maintained well above the top of the active fuel throughout the event.

Based upon the preceding analysis, it is concluded this event had no adverse impact on nuclear safety. The analysis is applicable to all power levels.

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CORRECTIVE ACTIONS

The main steam line isolation valve stem was replaced per Maintenance Work Order 2-01-03746. Local leak rate testing, valve cycling, and valve stroke timing were performed successfully and the valve was returned to an operable status.

Southern Nuclear will perform an investigation to determine the feasibility and cost of options to reduce or eliminate main steam line isolation valve stem assembly vibration.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

This LER does not contain any permanent licensing commitments.

Failed Component Information:

Master Parts List Number: 2B21-F028B EIIS System Code: SB
Manufacturer: Rockwell International Reportable to EPIX: Yes
Model Number: 1612 JM MNTY Root Cause Code: X
Type: Valve, Shutoff EIIS Component Code: SHV
Manufacturer Code: R344

Previous similar events in the last two years in which the reactor scrammed automatically while critical were reported in the following Licensee Event Reports:

50-321/2000-002, dated 2/25/2000
50-321/2000-004, dated 8/4/2000
50-321/2001-002, dated 5/21/2001
50-366/2001-002, dated 12/14/2001.

Corrective actions for these previous similar events could not have prevented this event because they involved different components and were the result of different causes.